RAIO-0418-59696

April 24, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 308 (eRAI No. 9261) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 308 (eRAI No. 9261)," dated December 22, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Question from NRC eRAI No. 9261:

- 12.02-3

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Steven Mirsky at 240-833-3001 or at smirsky@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9261

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NuScale Response to NRC Request for Additional Information eRAI No. 9261

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9261
Date of RAI Issue: 12/22/2017

NRC Question No.: 12.02-3

Regulatory Basis

10 CFR 52.47(a)(5) requires applicants to identify the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radiation exposures within the limits set forth in 10 CFR Part 20.

10 CFR 50.49 and 10 CFR Part 50, Appendix A, Criterion 4 require that certain components important to safety be designed to withstand environmental conditions, including the effects of radiation, associated with design basis events, including normal operation, anticipated operational occurrences, and design basis accidents. The Acceptance Criteria of DSRS section 3.11 “Environmental Qualification of Mechanical and Electrical Equipment,” states that the radiation environment should be based on the integrated effects of the normally expected radiation environment over the equipment’s installed life, plus the effects associated with the design-basis event during or following which the equipment is required to remain functional.

10 CFR 20.1101(b) and 10 CFR 20.1003, require the use of engineering controls to maintain exposures to radiation as far below the dose limits in 10 CFR Part 20as is practical. NuScale DSRS section 12.2 “Radiation Source,” regarding the identification of isotopes and the methods, models and assumptions used to determine dose rates. NuScale DSRS section 12.3 “Radiation Protection Design Feature,” states in the specific acceptance criteria that areas inside the plant structures should be subdivided into radiation zones, with maximum design dose rate zones and the criteria used in selecting maximum dose rates identified.

Background

DCD Tier 2 Revision 0 DCD Table 3C-1: “Environmental Qualification Zones - Reactor Building,” DCD Table 3C-8: “Accident EQ Radiation Dose,” and DCD Table 3C-6: “Normal Operating Environmental Conditions,” describe the integrated dose in and around the NuScale module containment vessel (CNV) due to normal operations and radiation exposure following an accident. DCD Section 12.2.1.13 “Post-Accident Sources,” states the post-accident source remains within the containment vessel except for assumed leakage into the bioshield envelope, which is above the surface of the pool water and below the bioshield. It further states that there are four volumes that are evaluated for the post-accident source term. Table 12.2-31 lists the integrated post-accident source energy deposition versus time for both photons and electrons for these four volumes. Table 12.2-28: “Post-Accident Source Term Input Assumptions,” list some of the assumptions used in the analysis. DCD Table 12.2-31: “Post-Accident Integrated Energy Deposition and Integrated Dose,” list some time integrated doses for periods following the accident. However, DCD Section 12.2 does not contain a table describing the airborne concentrations in the CNV or above the surface of the pool water and below the bioshield.

Key Issue 1:

Based on the review of material made available to the staff during the RPAC Chapter 12 Audit, the staff determined that the environmental qualification calculations performed to establish the post-accident total integrated dose (TID) between the top of the containment vessel (CNV) and the interior of the bioshield structure only considered accumulated dose due to gases released from the CNV following the accident. The methodology used by the applicant did not consider the additional contribution to TID due to gamma photons emanating from the gases in the CNV and the liquid in the CVN and reactor, which penetrate the CNV. Based on analysis by the staff, the additional TID associated with just the gases in the top of the CNV could be over a 1000 Rad/hr.

Key Issue 2:

Based on the review of material made available to the staff during the RPAC Chapter 12 Audit, the staff determined that the methodology used by the applicant to calculate the dose rate from gamma emissions used the total photon energy emission rate from individual isotopes, rather than the individual photon emission rate. Based on an analysis by the staff, this may underestimate doses from photons by over 20%.

Question

- Revise the calculations used to determine the post-accident dose rates to account for sources of radiation other than just the gas cloud above the surface of the pool water and below the bioshield,
- As necessary revise DCD Section 12.2.1.13 and DCD Table 12.2-28 to state the assumptions regarding radiation transport to and through the areas listed in Table 12.2-31,
- As necessary, revise the DCD radiation zone maps in DCD Section 12.3 to reflect the changes to the post-accident dose rates to areas affected by the shine from areas above the surface of the pool water and below the bioshield,
- As necessary, revise the thicknesses of shielding described in the DCD to reflect the increase in the photon strength resulting from the changes to the calculation method,
- As necessary, revise the EQ dose estimates and categories described in DCD Table 3C-1: “Environmental Qualification Zones - Reactor Building,” DCD Table 3C-8: “Accident EQ Radiation Dose,” and DCD Table 3C-6: “Normal Operating Environmental Conditions,” affected by the changes in photon strength caused the changes to the table in DCD section 12.2,

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

Key Issue 1:

A revised accident source term topical report (TR-0915-17565, Rev. 3) is scheduled to be issued in late summer 2018. This topical report will describe the methodology used to develop the revised NuScale accident source term. The reduced magnitude of the revised NuScale accident source term reduces the overall magnitude of the resulting post-accident doses. Using this revised accident source term, NuScale has performed calculations to determine the post-accident dose rates and integrated doses that account for sources of radiation from within the containment vessel (CNV) that shine into the bioshield envelope in addition to the sources of radiation that are released and leak from the NuScale power module (NPM) into the bioshield envelope.

FSAR Table 12.2-31 has been revised to incorporate changes to the total integrated doses for the regions associated with the CNV, reactor pressure vessel (RPV) and the bioshield envelope (see NuScale's response to RAI 12.2-11 (9268)).

For impacts to equipment qualification and FSAR Chapter 3, please refer to the NuScale response to RAI 8837.

Key Issue 2:

NuScale used an infinite cloud assumption for calculating the dose rates from gamma emissions. This assumption results in all the gamma photon energy that is emitted being deposited in the volume of interest, which is bounding for this analysis. This energy deposition calculation method is not influenced by discrete photon energies, as the total energy emitted for each nuclear transition is deposited at the location of release. Additionally, the shine contribution is not calculated based on average photon energy, but discrete energies, thus this modeling simplification does not result in underestimation of shine contributions.

Impact on DCA:

FSAR Section 12.2.1.13, Table 12.2-28, Table 12.2-29, Table 12.2-30, and Table 12.2-31 have been revised as described in the response above and as shown in the markup provided in this response.

12.2.1.11 Control Rods and Secondary Source Rods

Control Rod Assemblies

The control rod assemblies are irradiated during reactor operations. Because the reactor core operates in an all-rods-out configuration, it is assumed that only the tip of the control rod is irradiated. This portion of the control rod assembly (CRA) consists of Ag-In-Cd neutron absorber. The major input assumptions are listed in Table 12.2-25. The CRA gamma spectra are listed in Table 12.2-26.

Secondary Source Rod

The secondary source is antimony and beryllium (Sb-Be) and is irradiated for nine cycles. Flux is the same as for the in-core instruments (Section 12.2.1.10).

The gamma ray source strengths associated with the secondary source rods are listed in Table 12.2-27 for various times after shutdown.

12.2.1.12 Secondary Coolant System

The secondary coolant system is expected to contain minimal radioactivity during normal operations. Primary-to-secondary leaks through the steam generator can introduce primary coolant activity into the secondary system with the resultant contamination level being dependent upon the activity level in the primary coolant and the magnitude of the steam generator leak. Because the condensate polishing system is a full flow system, the condensate polishers were evaluated for the radioactive material that could accumulate on the resins during the period between resin regenerations. Assuming the secondary coolant is at the design basis concentrations (Table 11.1-5), resin decontamination factors consistent with NUREG-0017, and a ten day resin regeneration period, the accumulation of radioactive material is less than 100 mCi.

12.2.1.13 Post-Accident Sources

RAI 12.02-3, RAI 12.02-11

Consistent with 10 CFR 50.34(f)(2)(vii), areas that could contain post-accident sources were evaluated for equipment protection and access. ~~The accident source term used to evaluate equipment qualification is based on the core isotopic inventory presented in Table 11.1-1, using the methodology presented in Regulatory Guide 1.183. Because the design does not include post-accident recirculation piping outside containment, the post-accident source remains within the containment vessel except for assumed leakage into the bioshield envelope, which is above the surface of the pool water and below the bioshield. The radionuclide activity released from assumed fuel damage is assumed to be instantaneously and homogeneously released into containment.~~ The maximum primary coolant activity released from design basis accidents, that could be released into the bioshield envelope, is assumed to be released instantaneously and homogeneously throughout the bioshield envelope volume. The remaining primary coolant inventory is assumed to be released instantaneously and homogeneously throughout the containment atmosphere, where it shines and leaks into the bioshield envelope at the technical specification leak rate and remains in the envelope's volume for the duration of the event. Table 12.2-34 provides the maximum post-accident activity concentrations in the NPM on a mass basis. These concentrations apply to both the liquid and vapor spaces. Other major assumptions for the post-accident source term are listed in Table 12.2-28. ~~The release fractions and containment aerosol removal rates are shown in Table 12.2-29 and Table 12.2-30, respectively.~~ Plateout of activity onto containment surfaces is neglected due to the small containment volume and the lack of surface coatings inside containment. There is also no aerosol removal assumed. The containment air and water volumes are determined based on the reactor vessel being initially full of water and the reactor vessel and containment vessel water levels being in equilibrium. There are ~~four~~three volumes that are evaluated ~~for~~, which includes the post-accident source term. Table 12.2-31 lists the integrated post-accident source energy deposition versus time for both photons and electrons for these ~~four~~three volumes. Table 12.2-31 also tabulates the integrated doses for various times post-accident. For additional details on equipment qualification, see Section 3.11 and Appendix 3.C.

12.2.1.14 Other Contained Sources

There are no other identified contained sources that exceed 100 mCi, including HVAC filters. To evaluate the accumulation of radioactive material on the Reactor Building HVAC system HEPA filters, the airborne radioactivity in the Reactor Building due to pool evaporation and primary coolant leaks was deposited on filters assuming a 99 percent particulate efficiency and two years of operation. For the pool evaporation portion, the Reactor Building HVAC system provides a ventilation flow rate equivalent to one air volume change per hour. For the primary coolant leakage portion, the activity that becomes airborne is captured and filtered by the ventilation system. The resultant accumulation of radioactive material is less than 100 mCi.

COL Item 12.2-1: A COL applicant that references the NuScale Power Plant design certification will describe additional site-specific contained radiation sources that exceed 100 millicuries (including sources for instrumentation and radiography) not identified in Section 12.2.1.

12.2.2 Airborne Radioactive Material Sources

This section describes the airborne radioactive material sources that form part of the basis for design of ventilation systems and personnel protective measures, and also are considered in personnel dose assessment.

12.2.2.1 Reactor Building Atmosphere

Airborne radioactivity may be present in the RXB atmosphere due to reactor pool evaporation or primary coolant leakage. The airborne concentration is modeled as a buildup to an equilibrium concentration based on Bevelacqua (Reference 12.2-1) given the production rate and removal rate. The airborne concentration in the air space above the reactor pool is determined by using the peak reactor pool water source term. The input parameters are listed in Table 12.2-32.

RAI 12.02-3, RAI 12.02-11

Table 12.2-28: Post-Accident Source Term Input Assumptions

Parameter	Value
Containment release delay	~~4.9 hours~~0 hours
Containment release duration	~~1.5 hours~~1.0E-05 hours
Containment leak rate	0.2%/day
Containment leak rate after 24 hours	0.1%/day
Aerosol fraction of non-noble gases released	~~95~~100%
Bioshield envelope volume	~~6711~~6475 ft^3
Primary coolant water density	~~46.9~~43.6 lb/ft^3
Air density	0.07 lb/ft^3
Containment air volume	~~4549~~3635 ft^3
~~Reactor vessel air volume~~	~~1595 ft^3~~
Combined water volume	2500 ft^3

RAI 12.02-3, RAI 12.02-11

Table 12.2-29: ~~Post-Accident Core Inventory Release Fractions~~Not Used

~~Radionuclide Group~~	~~Release Fraction~~
~~Noble gases~~	~~0.68~~
~~Halogens~~	~~0.22~~
~~Alkali metals~~	~~0.24~~
~~Tellurium group~~	~~0.0057~~
~~Alkaline group~~	~~0.21~~
~~Molybdenum group~~	~~0.045~~
~~Noble metals~~	~~0.0011~~
~~Lanthanides~~	~~4.5E-08~~
~~Cerium group~~	~~4.5E-08~~

RAI 12.02-3, RAI 12.02-11

Table 12.2-30: ~~Post-Accident Containment Aerosol Removal Rates~~Not Used

~~Time after Core Damage (hours)~~	~~Removal Rate (hour^{-1})~~
~~4.9~~	~~48.6~~
~~5.94~~	~~6.9~~
~~6.94~~	~~4.03~~
~~7.94~~	~~5.92~~
~~8.94~~	~~7.84~~
~~22.82~~	~~34.3~~
~~36.71~~	~~68.3~~
~~50.59~~	~~100~~
~~64.48~~	~~100~~
~~78.36~~	~~0~~

RAI 12.02-3, RAI 12.02-11

Table 12.2-31: Post-Accident Integrated Energy Deposition and Integrated Dose

Volume	Medium	Time	~~Integrated Deposition (MeV)~~	Integrated Dose (Rad)
Reactor and containment	Water	1 hour	~~0.000E+00~~	~~0.000E+00~~2.73E+03
		~~End release~~	~~1.705E+21~~	~~5.136E+05~~
		~~1 day~~	~~1.872E+22~~	~~5.639E+06~~
		36 hour	~~2.709E+22~~	~~8.159E+06~~2.99E+04
		3 day	~~4.713E+22~~	~~1.420E+07~~4.01E+04
		30 day	~~2.492E+23~~	~~7.507E+07~~9.15E+04
		100 day	~~6.105E+23~~	~~1.839E+08~~1.17E+05
~~Reactor vessel~~	~~Air~~	~~1 hour~~	~~0.000E+00~~	~~0.000E+00~~
		~~End release~~	~~2.350E+21~~	~~7.435E+08~~
		~~1 day~~	~~2.471E+22~~	~~7.816E+09~~
		~~36 hour~~	~~3.549E+22~~	~~1.123E+10~~
		~~3 day~~	~~6.126E+22~~	~~1.938E+10~~
		~~30 day~~	~~2.896E+23~~	~~9.162E+10~~
		~~100 day~~	~~6.582E+23~~	~~2.082E+11~~
Containment vessel	Air	1 hour	~~0.000E+00~~	~~0.000E+00~~1.19E+06
		~~End release~~	~~2.350E+21~~	~~2.607E+08~~
		~~1 day~~	~~2.471E+22~~	~~2.741E+09~~
		36 hour	~~3.549E+22~~	~~3.936E+09~~1.35E+07
		3 day	~~6.126E+22~~	~~6.795E+09~~1.84E+07
		30 day	~~2.896E+23~~	~~3.213E+10~~4.85E+07
		100 day	~~6.582E+23~~	~~7.301E+10~~7.78E+07
Bioshield envelope	Air and water	1 hour	~~0.000E+00~~	~~0.000E+00~~7.35E+03
		~~End release~~	~~7.127E+16~~	~~5.359E+03~~
		~~1 day~~	~~5.078E+18~~	~~3.818E+05~~
		36 hour	~~1.033E+19~~	~~7.764E+05~~8.02E+04
		3 day	~~2.935E+19~~	~~2.207E+06~~1.11E+05
		30 day	~~4.792E+20~~	~~3.603E+07~~4.30E+05
		100 day	~~2.105E+21~~	~~1.583E+08~~1.30E+06

RAI 12.02-3, RAI 12.02-11

Table 12.2-34: Maximum Post-Accident Radionuclide Concentrations

Isotope	RCS Peak Concentration (µCi/g)
Kr-83m	2.69E+00
Kr-85m	2.14E-01
Kr-85	3.73E+01
Kr-87	6.52E-02
Kr-88	1.90E-01
Xe-131m	1.87E+00
Xe-133m	5.47E+00
Xe-133	2.07E+02
Xe-135m	1.73E+01
Xe-135	1.05E+02
Xe-137	1.39E-02
Xe-138	4.77E-02
Br-82	5.67E-01
Br-83	2.67E+00
Br-84	9.06E-01
Br-85	9.52E-02
I-130	4.45E+00
I-131	1.20E+02
I-132	4.60E+01
I-133	1.75E+02
I-134	2.09E+01
I-135	1.04E+02
Rb-88	3.31E-01
Rb-89	1.08E-02
Cs-134	1.84E-01
Cs-136	3.86E-02
Cs-137	1.27E-01
Cs-138	9.98E-02
Ni-63	4.41E-02
Sr-89	1.10E-02
Zr-95	6.51E-02
Nb-95	6.44E-02
Mo-99	3.88E-02
Tc-99m	7.01E-02
Ag-110m	2.17E-01
Te-132	1.56E-02
Ba-137m	2.26E-01
Na-24	9.11E+00
Cr-51	5.19E-01
Mn-54	2.67E-01
Fe-55	2.00E-01
Fe-59	5.01E-02
Co-58	7.68E+00
Co-60	8.83E-02
W-187	4.65E-01

Table 12.2-34: Maximum Post-Accident Radionuclide Concentrations (Continued)

Isotope	RCS Peak Concentration (μCi/g)
Zn-65	8.50E-02
H-3	9.90E-01
Ar-41	2.07E-01